Agencies and Persons Consulted

In accordance with its stated policy, the NRC staff consulted with the Arizona State official regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed exemption.

For further details with respect to this action, see the licensee's letter dated December 28, 1994, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Dated at Rockville, Maryland, this 24th day of March 1995.

For the Nuclear Regulatory Commission.

Theodore R. Quay,

Director, Project Directorate IV–2, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–7699 Filed 3–28–95; 8:45 am]

BILLING CODE 7590–01–M

Advisory Committee on Nuclear Waste; Notice of Meeting

The Advisory Committee on Nuclear Waste (ACNW) will hold its 73rd meeting on April 12–13, 1995, in Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The meeting will be open to public attendance, with the exemption of portions that may be closed to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(6).

The agenda for this meeting shall be as follows:

Wednesday and Thursday, April 12 and 13, 1995—8:30 A.M. until 6:00 P.M.

During this meeting the Committee plans to consider the following:

A. *DOE's Approach to Groundwater Travel Time*—The Committee will hear presentations by and hold discussions with representatives of the Department of Energy and the NRC staff and their consultants on the Department's approach to resolving the groundwater travel time issue associated with the proposed Yucca Mountain high-level waste repository.

B. *Meeting with the Director, NRC's Division of Waste Management, Nuclear Materials Safety & Safeguards*—The Director will provide information to the Committee on current waste management issues, which may include the use of expert judgment, a discussion of NRC's key technical uncertainties associated with Yucca Mountain, and a Performance Assessment Vertical Slice Review of volcanism and groundwater travel time.

C. *National Performance Review Phase 2*—The Committee will hear presentations by and hold discussions with the NRC staff on initiatives to streamline the federal government and regulatory process.

D. *Meeting with the Director, NRC's Office of Nuclear Materials Safety and Safeguards*—The Committee will welcome the new Director and discuss interactions between the Committee and the Office of NMSS.

E. *Preparation of ACNW Reports*—The Committee will discuss proposed reports on the Approach to Groundwater Travel Time at Yucca Mountain, a low-level waste branch technical position on performance assessment, a proposed NRC rule on radiological criteria for decommissioning and the EPA's preproposal standard on low-level waste disposal. Additional topics will be considered as time permits including the engineered barrier system for the proposed Yucca Mountain repository, the evaluation of rock mechanics for the proposed Yucca Mountain site, and DOE's program approach.

F. *Committee Activities/Future Agenda*—The Committee will consider topics proposed for future consideration by the full Committee and working groups. The Committee will also discuss organizational and personnel matters related to the selection of new ACNW members and ACNW staff. A portion of this session may be closed to public attendance to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(6).

G. *Miscellaneous*—The Committee will discuss miscellaneous matters related to the conduct of Committee activities and organizational activities and complete discussion of matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings were published in the **Federal Register** on October 7, 1994 (59 FR 51219). In accordance with these procedures, oral or written statements may be presented by members of the public, electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the ACNW Executive Director, Dr. John T. Larkins, as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for this purpose may be obtained by contracting the ACNW Executive Director prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACNW Executive Director if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the ACNW Executive Director, Dr. John T. Larkins (Telephone 301/415–7360), between 7:30 A.M. and 4:15 P.M. EST.

Dated: March 24, 1995.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 95–7737 Filed 3–28–95; 8:45 am]

BILLING CODE 7590–01–M

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 3, 1995, through March 17, 1995. The last biweekly notice was published on March 15, 1995.

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 28, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: February 24, 1995

Description of amendment request: The proposed change would remove Section 4.3 from the Technical Specifications (TS) because the primary system testing following opening is already performed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, as implemented in the licensee's inservice inspection program as required by TS 4.0.1.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

This change does not involve a significant hazards consideration for the following reasons.

1. The requested change does not involve a significant increase in the probability or consequences of an accident previously evaluated. This requested change will provide consistency between our Technical Specifications (TS) and 10 CFR 50.55a which requires testing in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. The requirements contained in TS Section 4.3 were placed into TS prior to incorporation of Section XI into the ASME Boiler and Pressure Vessel Code. The NRC and industry have since recognized the ASME Boiler and Pressure Vessel Code, Section XI as the appropriate testing program. Adequate assurance of primary system integrity will be provided since primary system testing will continue to be controlled and performed in accordance with the rules for inservice inspections provided by ASME Boiler and Pressure Vessel Code, Section XI as implemented by our approved In-Service Inspection (ISI) Program, as required by TS Section 4.0.1. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

2. The requested change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The requested change deletes the current TS requirements for primary system testing by recognizing that we will continue to perform required testing consistent with 10 CFR 50.55a and ASME Boiler and Pressure Vessel Code, Section XI, as implemented by our approved ISI Program, as required by TS Section 4.0.1. This requested change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. Therefore, the requested change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The requested change does not involve a significant reduction in a margin of safety. The requested change deletes the current TS Section 4.3 requirements for primary system testing and maintains the margin of safety by continuing to perform required testing in accordance with 10 CFR 50.55a and ASME Boiler and Pressure Vessel Code, Section XI, as implemented by our approved ISI Program, as required by TS Section 4.0.1. Therefore, the requested change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: William H. Bateman

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: March 3, 1995

Description of amendment request: The proposed amendment would eliminate the requirement to perform periodic measurement testing of the response times for selected pressure and differential pressure sensors. The requirement that reactor trip and engineered safety feature response time functions be within their specified limit at least once per 18 months will be verified instead of demonstrated. The associated bases section for response time requirements will be changed to allow the sensor response time portion of the channel response time to use historical records, testing results, or vendor supplied engineering specifications. No other changes to response time methods are included in this change.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not result in a condition where the design, material, or construction standards that were applicable prior to the change are altered nor does it modify any system interface. The same Reactor Trip System and Engineered Safety Features Actuation System instrumentation is being used; the time response allocations/modeling assumptions in the Final Safety Analysis Report (FSAR) Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed activity will not change, degrade, or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not alter the performance of the pressure and the differential pressure transmitters used in the plant protection systems. The sensors will still have response time verified by test before placing the sensor in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain sensors (assuring equipemt operable) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the sensor response characteristic. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

The proposed amendment to [sic] does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect sensor response time. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: William H. Bateman

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, IllinoisDocket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: May 20, 1994, as supplemented February 2, 1995

Description of amendment request: The proposed amendment would permit the licensee to use an alternate repair criteria (ARC), designated as the F* criteria. Use of the F* criteria would allow tubes with otherwise pluggable indications, to remain in service as long as the indications are below the designated minimum distance of the F* criteria. The F* criteria for Byron and Braidwood defines a length of 1.7 inches of undegraded expanded tube within the tubesheet as the minimum distance acceptable for implementing the ARC. Below the F* length, a circumferential tube defect can exist and the tube can remain in service. The proposed amendment will change the plugging limit definition and would exclude plugging steam generator tubes with indications that satisfy the F* criteria. The F* criteria maintains the structural integrity of the degraded tube as the primary pressure boundary and allows the tube to remain in service for heat transfer and core cooling.

This alternate repair criteria qualification is documented in Babcock & Wilcox Nuclear Technologies (BWNT) Topical Report BAW-10196 P Revision 1, ''W-D4 F* Qualification Report,'' which is included as part of the licensee's submittal. The staff's proposed no significant hazards consideration determination for the requested change was published on July 6, 1994 (59 FR 34659). In response to the staff's request for additional information by letter dated February 2, 1995, the licensee revised their previous submittal.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

A. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The supporting qualification report for subject criteria demonstrates that the presence of the tubesheet will enhance the tube integrity in the region of the tube-to-tubesheet roll expansions by precluding tube deformation beyond its initial expanded outside diameter. The resistance to a tube rupture is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into tubesheet provides a mechanical leak limiting seal between the tube and the tubesheet. A tube rupture cannot occur because the contact between the tube and the tubesheet does not permit sufficient movement of tube material.

The type of degradation for which the F* criteria has been developed (cracking with a circumferential orientation) can theoretically lead to a postulated tube rupture event provided that the postulated through-wall circumferential crack exists near the top of the tubesheet. An evaluation including analysis and testing has been done to determine the resistive strength of the expanded tubes within the tubesheet. This evaluation provides the basis for the acceptance criteria for tube degradation subject to the F* criteria. The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. For consistency with current offsite dose limits, the site allowable leakage limit during a MSLB has been conservatively calculated to be 12.8 gpm for Byron and 9.1 gpm for Braidwood, which includes the accident leakage from IPC in addition to the accident leakage from F* on the faulted steam generator and the operational leakage limit. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F*. As a requirement for operation following application IPC, the projected distribution of crack indications over the operating period must be verified to result in primary to secondary accident leakage less than the site allowable leakage limit. Thus, the consequences of a MSLB remain unchanged.

The tube rupture and pullout is fully bounded by the existing steam generator tube rupture analysis included in the UFSAR. The leakage testing of the roll expanded tubes indicates that for tube expansion lengths approximately equal to the * distance, any postulated primary to secondary leakage from * tubes would be insignificant. The proposed alternate repair criteria does not adversely impact any other previously evaluated design basis accident.

The leakage from an F* tube would be limited by the tube-to-tubesheet interface since this leak would occur below the secondary face of the tubesheet. Qualification testing and previous experience indicate that normal and faulted leakage is well below Technical Specification and administrative limits creating no increase in the consequences associated with tube rupture type leakages. The UFSAR analyzed accident scenarios are still bounding since the normal and faulted leak rates are well within the normal operating limit of 150 gallons per day. This conclusion is consistent with previous F* programs approved and used at other operating plants.

All of the design and operating characteristics of the steam generator and connected systems are preserved since the F* criteria utilizes the ''as rolled'' tube configuration that exists as part of the original steam generator design. The F* joint has been analyzed and tested for design, operating, and faulted condition loadings in accordance with Regulatory Guide 1.121 safety factors. The potential for a tube rupture is not increased from the original submittal as demonstrated in the qualification analyses and testing completed in the BWNT report.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

Implementation of the proposed F* criteria does not introduce any changes to the plant design basis. Use of the criteria does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. In the unlikely event the failed tube severed completely at a point below the F* region, the remaining F* joint would retain engagement in the tubesheet due to its length of expanded contact within the tubesheet bore. This engagement length would prevent any interaction of the severed tube with neighboring tubes. Any hypothetical accident as a result of any tube degradation in the expanded region of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leak tightness will be maintained such that any postulated accident leakage from F* tubes will be negligible with regard to offsite doses.

Therefore, there is not a potential for creating the possibility of a new or different type of accident from any accident previously evaluated.

C. The proposed changes do not involve a significant reduction in a margin of safety.

The use of the F* criteria has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Regulatory Guide 1.121 and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the * criteria is any degradation indication in the tubesheet region, more than the F* distance from the secondary face of the tubesheet or the top of the last hardroll contact point whichever is further into the tubesheet. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code and Regulatory Guide 1.121 used in steam generator design. The * distance has been verified by various testing to be greater than the length of the roll expanded tube-to-tubesheet interface required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. The protective boundaries of the steam generator continue to be maintained with the use of the F* criteria. A tube with the indication of degradation previously requiring removal from service can be kept in service through the F* criteria. Since the joint is contained within the tubesheet bore, there is no additional risk associated with the previously analyzed tube rupture event. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the UFSAR accident analyses.

Implementation of the alternate repair criteria will decrease the number of tubes which must be taken out of service with tube plugs or repaired by sleeves. Both plugs and sleeves reduce the RCS flow margin; thus, implementation of the F* criteria will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

NRC Project Director: Robert A. Capra

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: November 22, 1994, as supplemented January 30, March 2, and March 13, 1995.

Description of amendment request: This request was previously published in the Federal Register on February 15, 1995 (60 FR 8746). It is being renoticed to provide clarification to the scope of the original request. The amendments would revise Technical Specification (TS) 3.8 to establish restricted loading patterns and associated burnup criteria for placing fuel in the Oconee spent fuel pools. In addition, the Design Features sections associated with the reactor and fuel storage would be revised. These changes are necessary to address two new fuel designs which have increased initial fuel enrichment and therefore cannot be stored in the spent fuel pools under existing TS or loaded into the reactor. An administrative change would be made to TS 6.9.1 to include spent fuel pool boron concentration in the Core Operating Limits Report. Other administrative changes would be made in the Design Features section to make the specification consistent with wording in the standard TS. Finally, the two additional supplements to the original request are referenced herein.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Standard 1. The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 15 parameters to determine the effect of the Cycle 16 reload and to ensure that the acceptance criteria of the FSAR safety analyses remain satisfied. The transient evaluation of Cycle 16 is considered to be bounded by previously accepted analyses. Section 7 of the Reload Report addresses ''Accident and Transient Analysis'' for this core reload.

There is no increase in the probability or consequences of an accident due to the spent fuel storage restrictions proposed in this amendment request. It has been shown that the calculated, worst case k_{eff} for this area is [less than or equal to] 0.95 under all conditions. There is no increase in the probability of a fuel drop accident in the SFP [spent fuel pool] since the mass of the new assemblies is not significantly different from the mass of the old assemblies. The likelihood of other accidents, previously evaluated and described in the FSAR, is also not affected by the proposed changes. In fact, it could be postulated that since the increase in fuel enrichment will allow for extended fuel cycle lengths, there will be a decrease in fuel movement and the probability of an accident may actually be reduced. There is also no increase in the consequences of a fuel rod drop accident in the SFP since the fission product inventory of individual fuel assemblies will not change significantly as a result of increasing the initial enrichment. In addition, no change to safety related systems is being made. Therefore, the consequences of a fuel rupture accident remain unchanged. In addition, it has been shown that K_{eff} all conditions. Therefore, the consequences of a criticality accident in the SFP remain unchanged as well. The above analysis ensures that the proposed reload amendment request will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses performed in support of this reload are in accordance with the NRC approved methods delineated in Specification 6.9.2. The predicted operating characteristics of Oconee 3 Cycle 16 are similar to previously licensed designs. The Mark B10T and Mark B11 fuel assembly designs remain mechanically compatible with all fuel handling equipment. Therefore, no new or different kind of fuel handling accident is created by the proposed amendment request.

Section 15.11 of the Oconee FSAR states that the refueling boron concentration is maintained such that a criticality accident during refueling is not considered credible. The proposed amendment request continues to assure that a criticality accident in the SFP or during refueling is not credible. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, by requiring a minimum boron concentration in the SFP, a criticality accident caused by violating the SFP storage restrictions is not considered credible. Therefore, the proposed amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The Oconee 3 Cycle 16 design was performed using the NRC approved methods given in Specification 6.9.2. The safety limits for Oconee 3 Cycle 16 are unchanged from previous cycles. The limits and margins summarized in the Oconee 3 Cycle 16 Reload Report are well within the allowable limits and requirements, and reflect no reductions to any margins of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Entergy Operations Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: December 14, 1993, as supplemented by letter dated March 3, 1995.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) by removing the reactor vessel material specimen withdrawal schedule and by updating the reactor coolant system pressure-temperature (P-T) curves. The specimen withdrawal schedule will be relocated to the Updated Final Safety Analysis Report (UFSAR). The original Notice was published on January 19, 1994 (59 FR 2867).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Although the Reactor Vessel material specimens withdrawal schedule will be removed from the Technical Specifications, the Technical Specifications bases will continue to provide background information on the use of the data obtained from material specimens. Also, updates to the schedule will continue to be submitted to the NRC for approval prior to implementation.

Operating the plant in accordance with the new, updated P-T Curves will assure preserving the structural integrity of the reactor vessel over the life of the plant. The pressure and temperature limits were developed in accordance with 10 CFR [Part] 50 Appendix G requirements.

Removing the requirements associated with the previous exemption to Appendix H (TS 4.4.8.1.2 items a & b) is purely an administrative change.

Therefore, the proposed changes will not significantly increase the probability or consequences of any accident previously evaluated.

Removal of the Reactor Vessel material specimen schedule from the Technical Specifications has no impact on accidents at the plant. Updates to the schedule will still be required to be submitted to the NRC prior to implementation per Section II.B.3 of Appendix H to 10 CFR Part 50.

Also, updates to the P-T Curves will not create a new or different type [of] accident. The reactor vessel beltline P-T limits were revised applying the general guidance of the ASME Code, Appendix G procedures with the necessary margins of safety for heatup, cooldown and inservice hydro test conditions.

The change to TS 4.4.8.1.2 items a & b is purely administrative.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Removal of the schedule for Reactor Vessel material specimen withdrawal from the Technical Specifications does not impact the margin of safety. The schedule will continue to receive NRC review and approval prior to implementation of updates to the schedule.

Updates to the P-T Curves are provided to preserve the margin to [sic] safety to assure that when stressed under operating, maintenance and testing the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The change to TS 4.4.8.1.2 items a & b is purely administrative.

Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: February 22, 1995

Description of amendment request: The proposed changes are administrative in nature in that reference to an ''automatic'' containment air lock tester will be deleted from TS 4.6.1.3. The automatic airlock tester is no longer being used.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment is administrative in nature in that the revision will eliminate the wording associated with optional use of the personnel airlock automatic leakage tester. The requirement for testing the personnel airlock at a pressure greater than or equal to P_a for at least 15 minutes remains unchanged. The acceptance criteria of personnel airlock seal leakage less than 0.01 L_a is also unchanged. The automatic leakage tester is not an accident initiator nor a part of the success path(s) which function to mitigate accidents evaluated in the plant safety analyses. The proposal does not involve any changes to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does it alter any assumptions or conditions in the plant safety analyses. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2)Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment to remove the reference to the personnel airlock automatic tester from the technical specifications will not introduce any new failure modes or system interactions, nor will it require the installation of any new or modified equipment. The requirement to leak test the personnel air locks will not be changed. Thus, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3)Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment is administrative in nature in that it eliminates the reference to the personnel airlock automatic leakage tester but does not alter the surveillance and acceptance criteria for such testing. Seal leakage testing is performed in accordance with an approved plant procedure which allows use of either an automatic tester or a portable testing cart. The automatic leakage tester is not used to actuate safety related equipment, provide interlocks, or perform plant control functions. The conditions evaluated in the plant accident and transient analyses do not involve this tester. The proposed change does not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: February 27, 1995

Description of amendment request: The proposed amendment will change Table 3.3-3 and 3.3-4 to accommodate an improved coincidence logic and relay replacement for the 4.16 kV Loss of Voltage Relays. Actions required for certain trip units with the number of operable channels one less than the total number of channels will also be changed. In addition, the format used to state the time delay for the 4.16 kV Degraded Voltage trip unit will be revised.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1)Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will result in a better overall posture of the plant under degraded/loss of voltage conditions. The design upgrade for the 4.16 kV Loss of Voltage system is more reliable, has inherently higher accuracy, and is easier to maintain and calibrate in the field. The coincidence logic will eliminate the spurious plant trip potential from the existing design. Restating the maximum time delay for the 4.16 kV Degraded Voltage (coincident with SIAS [safety injection actuation signal]) protective relays in a ''less than'' format will assure that the transfer of power to the on-site sources occurs before the level of voltage becomes injurious to the equipment under accident conditions, and will ensure that stripping of the emergency power busses and loading of the EDG (s) [emergency diesel generators] will occur within the time allowed by original design criteria. The maximum allowed time delay for this function is not being increased, and the time delay assumed in the accident analyses for connecting the emergency bus to the diesel generator will not be exceeded. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2)Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not change the operation, function or modes of plant operation. The ability of the loss of power and degraded grid voltage protection scheme to properly transfer from the off-site to the on-site power sources is being maintained. The relays in the improved design of the 4.16 kV Loss of Voltage function are of the type presently being used in identical applications at both St. Lucie plant units. No new hazards are created or postulated which may cause an accident different from any accident previously analyzed. The modifications will result in a more sensitive protection scheme allowing continuous operation without unnecessary challenges to the safety systems, and will continue to provide adequate protection to all the safety equipment. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3)Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The capability of the loss of power and degraded grid voltage protection scheme is enhanced by the changes being proposed and is confirmed by the existing surveillance requirements. The planned modifications to the 4.16 kV Loss of Voltage function will result in a more sensitive undervoltage detection system and reduce the possibility of spurious actuation. The maximum time assumed in the safety analyses for connecting each Emergency Bus to its dedicated Emergency Diesel Generator is not being changed, and assurance that separation from a degraded off-site power source will occur before this time interval is exceeded during accident conditions will be maintained by the proposed amendment. Accordingly, the margin of safety is not affected. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting Evaluation of Proposed TS [Technical Specifications] Changes, FPL has concluded that this proposed license amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: February 27, 1995

Description of amendment request: The proposed amendment will modify surveillance requirement (SR) 4.9.8.1 and 4.9.8.2 to allow a reduction in the required minimum shutdown cooling flow rate under certain conditions during operational MODE 6. In addition, the format of the SR will be changed to clarify the intent of the stated surveillances.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Operation of the SDCS [shutdown cooling system] is not an accident initiator and, therefore, does not significantly increase the probability of an accident previously evaluated.

The proposed change will allow a plant configuration needed to perform maintenance activities on LPSI [low-pressure safety injection]/SDCS headers by isolating one injection flow line for an operable SDCS train during certain MODE 6 conditions. In the event of a failure or unavailability of the alternate SDCS train, this configuration could result in the proposed minimum flow rate.

The proposed change only modifies the minimum required flow rate, and does not affect the probability of this event. FPL has evaluated the proposed value of reactor coolant flow and has shown that the bases for the existing LCO [limiting condition for operation] will continue to be satisfied. Therefore, there are no significant increases in the consequences of any event from the proposed change. No other system interactions are involved related to previously evaluated accidents, and the proposed change has no adverse effect on any other system performance.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect the normal operation of the plant. No new systems are introduced and there is no adverse effect on any other system configuration or performance. The change will, however, allow isolation of one SDCS injection flow path for maintenance activities in MODE 6 under controlled conditions. The failure of the alternate SDCS train does not create a new accident and has been further evaluated in the reduced flow configuration, and shown to meet all the TS bases requirements. Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3)Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The safety considerations related to the proposed change are described in the bases to TS [Technical Specification] 3/4.9.8. FPL has evaluated the proposed reduction in SDCS flow requirement, under stated conditions, and has shown that the proposed flow rate meets all the TS bases requirements involving decay heat removal, boron dilution, and stratification. Established acceptance criteria providing margins of safety are not being changed by the proposed amendment. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting Evaluation of Proposed TS Changes, FPL has concluded that this proposed license amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:*Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: March 14, 1995

Description of amendment request: Georgia Power Company (GPC or the licensee) has proposed a temporary change to Hatch Unit 2 Technical Specification (TS) Required Action 3.3.6.1.F.1, and associated Bases. The proposed change would add a note to the Primary Containment Isolation Instrumentation actions to permit the drywell and wetwell purge valves which are isolated by the drywell radiation monitor signal to be opened with one inoperable drywell radiation monitor. The note will expire prior to startup from the Hatch Unit 2 refueling/maintenance outage scheduled in the fall of 1995, at which time the radiation monitor can be repaired or replaced. Should the unit be forced into a cold shutdown of sufficient duration (i.e., drywell de-inerted), the inoperable radiation monitor will be repaired at that time. The TS containment sections allow these valves to be opened for inerting, de-inerting, and pressure control. However, with radiation monitor 2D11-K621B inoperable, the primary containment isolation instrumentation TS require the valves be closed until the unit achieves a cold shutdown condition. Without the ability to open these valves until cold shutdown, pressure control and de-inerting are difficult.

The purpose of the high drywell radiation primary containment isolation signal is to limit fission product release following a postulated loss-of-coolant accident (LOCA) with significant fuel damage. It is one of several signals which isolate the primary containment vent and purge valves. A high drywell pressure signal will not only shut down the reactor and generate a LOCA signal, it will also isolate these valves.

High drywell radiation indicates possible gross failure of the fuel cladding. The generation of this isolation signal is not credited in any accident or transient analysis. Chapter 15 of the Hatch Unit 2 Final Safety Analysis Report (FSAR) discusses the radiological consequences of a postulated large break LOCA with fuel failure to show conformance to 10 CFR Part 100 and 10 CFR Part 50, Appendix A. This analysis is not affected.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

The change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Opening the containment purge and vent valves with an inoperable drywell radiation monitor will not increase the probability of any previously evaluated accident. The fact that the monitor cannot send an automatic isolation signal will not significantly affect the consequences of an accident. The function of the primary containment isolation signal is to detect and limit release of fission products following significant fuel damage. The generation of this isolation signal is not credited in any accident or transient analysis. Chapter 15 of the Unit 2 FSAR evaluates the radiological consequences of a postulated design basis LOCA with non-mechanistic fuel damage. This licensing evaluation shows conformance to the radiological limits presented in 10 CFR 100 and 10 CFR 50, Appendix A. The results of this analysis are not affected since the valves are otherwise operable and receive isolation signals from other instrumentation.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment designed to prevent or mitigate the consequences of accidents or transients. Therefore, the change has no effect on any accident initiator, and no new or different type of accidents are postulated to occur.

3. The proposed amendment does not result in a significant reduction in the margin of safety.

As discussed in Item 1 above, the assumptions and results of the licensing evaluations remain unchanged. Therefore, the margin of safety is not significantly affected.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:*Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Herbert N. Berkow

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: February 28, 1995

Description of amendment request: Technical Specification (TS) Section 6.5.1.12 would be revised to delete the requirement to render determinations in writing with regard to whether or not activities listed in TS Sections 6.5.1.2 and 6.5.1.5 constitute an unreviewed safety question.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed change removes the requirement to render determinations in writing with regard to whether or not opposed changes to the Technical Specifications and investigations of violations of Technical Specifications constitute an unreviewed safety question. This change is considered an administrative change to remove a requirement which is not relevant to these activities and which is also consistent with the BWR Revised Standard Technical Specifications (NUREG 1433). Existing requirements to perform Technical and Independent Safety Reviews of these activities are not affected. Therefore, the proposed amendment does not significantly increase the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change is considered administrative since it removes a requirement which is not relevant to the affected activities, and which is also consistent with the BWR Revised Standard Technical Specifications Administrative Controls for Review and Audit. Existing requirements to perform Technical and Independent Safety Reviews for the affected activities are not changed. Therefore, this change has no effect on the possibility of creating a new or different king of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed change removes a requirement which is not relevant to the affected activities. Existing Technical Specification requirements to perform Technical and Independent Safety Reviews for the affected activities are not changed and therefore, will continue to ensure that such activities properly address nuclear safety and safe plant operation. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:*Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Phillip F. McKee

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: February 15, 1995

Description of amendment request: The proposed amendment would modify (by relocation to the Technical Requirements Manual) Technical Specification (TS) 3/4.3.3.7, Chemical Detection Systems, and TS 3/4.8.4.1, Electrical Equipment Protective Devices - Containment Penetration Conductor Overcurrent Protective Devices, and the associated Bases.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change to Technical Specification 3.3.3.7, Chemical Detection Systems and 3.8.4.1, Electrical Equipment Protective Devices-Containment Penetration Conductor Overcurrent Protective Devices, is of an administrative nature in that the listed Technical Specifications and Bases will be relocated in entirety to the Technical Requirements Manual (TRM). Any future changes to the relocated requirements will be in accordance with 10CFR 50.59 and approved station procedures. Whether the listed Technical Specifications and Bases are located in Technical Specifications or the Technical Requirements Manual has no effect on the probability or consequences of any accident previously evaluated.

The proposed change does not alter the assumptions previously made in the listed Technical Specifications. The proposed change allows the Commission and South Texas more effective use of personnel resources to control requirements that meet the four Criteria in the Final Policy Statement. The proposed change will not change the dose to workers.

Since the probability of a [sic] accident is unaffected by the administrative relocation of the listed Technical Specifications, and the doses are not affected and do not exceed acceptance limits, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change to Technical Specification 3.3.3.7, Chemical Detection Systems and 3.8.4.1, Electrical Equipment Protective Devices-Containment Penetration Conductor Overcurrent Protective Devices, is of an administrative nature in that the listed Technical Specifications and Bases will be relocated in entirety to the Technical Requirements Manual (TRM). Any future changes to the relocated requirements will be in accordance with 10CFR 50.59 and approved station procedures. Whether the listed Technical Specifications and Bases are located in Technical Specifications or the Technical Requirements Manual has no effect on any previously evaluated accident. It does not represent a change in the configuration or operation of the plant and, therefore, does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

The proposed change to Technical Specification 3.3.3.7, Chemical Detection Systems and 3.8.4.1, Electrical Equipment Protective Devices-Containment Penetration Conductor Overcurrent Protective Devices, is of an administrative nature in that the listed Technical Specifications and Bases will be relocated in entirety to the Technical Requirements Manual (TRM). Any future changes to the relocated requirements will be in accordance with 10CFR 50.59 and approved station procedures. The margin of safety is not reduced when the requirements are relocated to a Licensee-controlled document because the requirements to change a License Basis Document via the 10CFR 50.59 process ensures the same questions concerning the margin to safety required for a License Amendment are asked. The major difference is the time and expense required for the License Amendments. Therefore, this proposed change does not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

NRC Project Director: William D. Beckner

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: February 15, 1995

Description of amendment request: The proposed amendment would modify Technical Specification 4.6.2.3.a.2 (and associated Bases) to reflect the reactor containment fan cooler flow rate assumed in the accident analyses and to specify that this flow is provided by the component cooling water system.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change to Technical Specification 4.6.2.3.a.2 is to reflect the cooling water temperature assumed in the accident analyses. The revised Technical Specification surveillance requirement will change the cooling water flow rate requirement to each Reactor Containment Fan Cooler from greater than or equal to 550 gallons per minute to greater than or equal to 1800 gallons per minute.

The proposed change, which will result in an increased acceptance criteria for the flow to the Reactor Containment Fan Coolers, is not indicative of accident initiators. The change will ensure that the surveillance requirement reflects the flow rate value assumed in the South Texas Project accident analyses and that the design and operability requirements of equipment important to safety are ensured.

The accident mitigation features of the plant are not affected by the proposed change since the change reflects the original assumptions made in the design of the accident mitigation features of the South Texas Project. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not create the possibility of a new or different kind of accident previously evaluated in the Safety Analysis Report because all the accidents were analyzed with a flow rate of 1800 gallons per minute to the Reactor Containment Fan Cooler.

3. Does the proposed change involve a significant reduction in a margin of safety?

There will be no adverse affects on margins of safety since a more stringent surveillance requirement will be applied to the Reactor Containment Fan Cooler. The Technical Specification operability and surveillance requirements are not reduced but rather made more restrictive by this proposed change. The change ensures that the margin of safety originally intended for the Reactor Containment Fan Coolers is maintained.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

*Local Public Document Room location:*Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

NRC Project Director: William D. Beckner

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: February 13, 1995

Description of amendment request: The proposed amendment would delete the audit frequency requirements from the Duane Arnold Energy Center Technical Specifications (TS) and add them to the Quality Assurance Program Description located in the Updated Final Safety Analysis Report.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) The proposed amendment does not involve a change in the probability or consequences of an accident previously evaluated. No physical changes will occur as a result of this amendment. The change is administrative in nature and does not impact the operation of the plant or the plant's response to any accident. Because it will allow management the flexibility to adjust the audit frequencies based upon the performance of the program or organization being audited, the overall performance of the organization will be improved.

(2) The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. No physical changes will occur as a result of this amendment. The change is administrative in nature and does not affect the operation or design of the plant; therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated. The audits will continue to be performed to provide assurance of conformance to the applicable requirements.

(3) The proposed amendment will not reduce the margin of safety. No physical changes will occur as a result of this amendment. The change is administrative in nature and does not affect the operation or design of the plant. Safety limits and limiting safety system settings are not affected by this proposed amendment. The amendment removes requirements for frequency of audits from the TS, thus permitting more effective scheduling of audits based on performance and the status of the activities audited. This should result in a more effective audit program that will contribute to an improvement in the overall performance of the organization.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis & Bockius, 1800 M Street, N. W., Washington, D. C. 20036-5869NRC Acting Project Director: John N. Hannon

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: February 10, 1995

Description of amendment request: The proposed amendment would modify Technical Specification 3.3.2.1, ''Control Rod Block Instrumentation,'' to revise two surveillance requirements and their associated notes for the Rod Withdrawal Limiter (RWL) mode of the Rod Pattern Control System. These changes will conform these requirements to their original bases and eliminate the potential for unnecessary power reductions.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

(1) The proposed changes are consistent with the Rod Withdrawal Error (RWE) analysis presented in Clinton Power Station (CPS) Updated Safety Analysis Report (USAR) Section 15.4.2. The proposed changes do not result in any change to plant equipment or operation; only the plant conditions for which the Rod Withdrawal Limiter (RWL) function(s) are required to be tested are being revised. The proposed changes continue to ensure that the RWL is OPERABLE and tested to ensure that continuous control rod withdrawals remain within the assumptions of the RWE analyses. The proposed changes have no impact on the probability of occurrence of a RWE event. Therefore, the proposed changes do not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed changes do not result in any changes to plant equipment or operation; only the plant conditions for which the RWL

function(s) are required to be OPERABLE and tested are being revised. The proposed changes continue to ensure that the RWL is OPERABLE and tested to ensure that continuous control rod withdrawals remain within the assumptions of the RWE analyses. As a result, no new failure modes are introduced. The proposed changes are clearly within the limits of plant operation as described in the USAR and the RWE analyses. Therefore, the proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed changes revise the testing requirements to be consistent with the testing required prior to Amendment No. 95. The proposed changes ensure that the RWL is OPERABLE and tested to ensure that continuous control rod withdrawals remain within the assumptions of the RWE analyses. The proposed changes are clearly within the limits of plant operation as described in the USAR and the RWE analyses. Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:*Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th St., Decatur, Illinois 62525.

NRC Acting Project Director: John N. Hannon

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine YankeeAtomic Power Station, Lincoln County, Maine

Date of amendment request: February 14, 1995

Description of amendment request: The proposed amendment would change responsibility for audits of the emergency and security plans and their implementing procedures. Audit responsibility would change from the licensee's Nuclear Safety Audit and Review (NSAR) Committee and the Plant Operation Review Committee (PORC), to the respective emergency and security plans. The proposed amendment is consistent with the guidance of NRC Generic Letter 93-07, Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans, dated December 28, 1993.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the Standards of 10 CFR 50.92(c). A summary of the licensee's analysis is presented below:

1. The proposed amendment would not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes do not have a direct effect on the physical plant or the maintenance of the physical plant, but would improve the safe operation of the plant by reducing the administrative burden of PORC and NSAR. This change would allow a better focus of management resources to the operational safety oversight of plant activities. The requirement to review, audit, document, control, and submit for regulatory review, the Emergency Plan and the Security Plan and their implementing procedures, is defined by regulation and remains unchanged. The proposed changes will not, of themselves, result in any reduction in the effectiveness of either the Emergency Plan or the Security Plan to protect the health and safety of the public. The proposed changes, therefore, will not increase the probability or consequences of an accident previously evaluated.

2. The proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

This change is administrative in nature and does not change or modify the physical plant or maintenance of the physical plant. Applicable regulations continue to enforce the requirement for review and audit by individuals not responsible for implementation of the existing programs. Consequently, independent oversight of the programs and procedures is not compromised by these proposed changes. The possibility of a new or different accident from any previously evaluated as a result of future changes in the implementation of the Security or Emergency Plans is not created.

3. The proposed amendment would not involve a significant reduction in a margin of safety.

The proposed changes will revise the administrative responsibilities of the PORC and NSAR committees allowing a better focus of resources on operational safety reviews. The requirements of the applicable Federal and State regulations ensure the continued effective oversight of the implementation of Security and Emergency Plans. Consequently, the adoption of the proposed changes would not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, Maine 04011

NRC Project Director: Walter R. Butler

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: October 18, 1994, as supplemented February 21, 1995.

Description of amendment request: The following changes requested in the October 18, 1994, submittal were published in **Federal Register** on November 9, 1994 (59 FR 35876). The proposed amendment would require three Type A overall Integrated Containment Leakage Tests be conducted at approximately equal intervals during shutdowns during each 10 year service period. For the third Type A test for the second 10-year period, it would be conducted during the thirteenth refueling outage extending the second 10-year service period to the end of the thirteenth refueling outage. The amendment would also change the Containment Leakage Bases by reflecting the conditions of a proposed exemption to 10CFR50, Appendix J, that would remove the requirement that the third Type A test for each 10-year period be conducted when the plant is shutdown for the 10-year plant inservice inspection.

By letter dated February 21, 1995, the licensee withdrew the action related to conducting the third Type A test for the second 10-year period during the thirteenth refueling outage and the reference to a proposed exemption to 10 CFR 50, Appendix J, that would remove the requirement that the third Type A test for each 10-year period be conducted when the plant is shutdown for the 10-year plant inservice inspection. The following basis for the proposed no significant hazards consideration determination relates to the February 21, 1995, request.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change does not involve a SHC because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

Type A tests are performed to ensure that the total leakage from containment does not exceed the maximum allowable primary containment leakage rate at the design pressure. This assures compliance with the dose limits of 10CFR100.

The proposed change to Surveillance Requirement 4.6.1.2.a of the Millstone Unit No. 2 Technical Specifications will increase the flexibility for scheduling the Type A tests. It does not modify the maximum allowable leakage rate at the design containment pressure, does not impact the design basis of the containment, and does not make any physical or operational changes to existing plant structures, systems, or components.

Historically, Type A tests have a relatively low failure rate where Type B and C testing (local leakage rate tests) could not detect the leakage path. Most Type A test failures are attributed to failures of Type B or C components (containment penetrations and isolation valves). Type B and C components are tested per Surveillance Requirement 4.6.1.2.d of the Millstone Unit No. 2 Technical Specifications. These tests are required to be conducted at intervals no greater than 24 months. These local leakage rate tests provide assurance that containment integrity is maintained. The Type B and C tests will continue to be performed in accordance with the requirements of Surveillance Requirement 4.6.1.2.d.

The previous Type A, B, and C tests demonstrate that Millstone Unit No. 2 has maintained control of containment integrity by maintaining a conservative margin between the acceptance criterion and the ''As-Found'' and ''As-Left'' leakage results. Based on this, the Millstone Unit No. 2 containment is considered to be in sound condition.

Based on the above, the proposed change to Surveillance Requirement 4.6.1.2.a of the Millstone Unit No. 2 Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change to Surveillance Requirement 4.6.1.2.a of the Millstone Unit No. 2 Technical Specifications will increase the flexibility in scheduling the Type A tests. It does not make any physical or operational changes to existing plant structures, systems, or components. In addition, the proposed change does not modify the acceptance criteria for the Type A tests. Maintaining the leakage through the containment boundary to the atmosphere within a specific value ensures that the plant complies with the requirements of 10CFR100. The containment boundary serves as an accident mitigator; it is not an accident initiator. Therefore, the proposed change to Surveillance Requirement 4.6.1.2.a does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

The proposed change to Surveillance Requirement 4.6.1.2.a of the Millstone Unit No. 2 Technical Specifications will increase the flexibility for scheduling the Type A tests. It does not modify the maximum allowable leakage rate at the design containment pressure, does not impact the design basis of the containment, and does not make any physical or operational changes to existing plant structures, systems, or components.

Based on the above, the proposed change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: February 1, 1995

Description of amendment request: This Technical Specification (TS) change would modify the applicable operational conditions for the secondary containment isolation radiation monitors located on the refueling floor and for the radiation monitor located in the railroad access shaft. Specifically, for the refueling floor exhaust duct and wall exhaust duct radiation monitors, the proposed change would modify the applicable operational condition during specific control rod testing evolutions which are core alterations and would indicate that the operability requirement does not apply during shutdown margin demonstrations. For the railroad access shaft exhaust duct radiation monitor, the change to the TS would modify the applicable operational condition to address plant evolutions involving irradiated fuel transfer within the railroad accessshaft and above the access shaft with the equipment hatch open.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

a. The proposed change to the applicable operational condition for the refueling floor process radiation monitors does not affect the probability of the design basis accidents. The monitors function in response to an airborne radioactivity concentration in the unfiltered air from the Zone III exhaust system and provide isolation signals which limit offsite doses to within regulatory limits. As such, there is no correlation between monitor operability and accident probability. The monitors act to mitigate the offsite effects of airborne contamination producing accidents, they are not potential accident initiators.

The proposed change does not result in a significant increase in the consequence of the design basis accidents. The postulated event associated with control rod related CORE ALTERATIONS which could result in increased Zone III airborne radioactivity concentrations is criticality resulting from a single control rod withdrawal, resulting in release of fission products. The probability of an unintended criticality from a single control rod withdrawal is low, and the potential for this criticality to result in fuel failure under shutdown conditions is even more remote. Withdrawal of a single control rod is an analyzed evolution during which time adequate design and operating controls exist to preclude criticality. However, in the unlikely event criticality should occur, the potential offsite effects would not be significant. Localized criticality involving a leaking rod, or criticality induced fuel failure, are the postulated mechanisms by which an increase in Zone III airborne radioactivity could be attained. Neither of these postulated, but very unlikely events, will result in radioactive release in excess of 10CFR100 limits. Any release would be monitored by instrumentation in the Reactor Building vent stack required to be OPERABLE at all times. In addition, Area Radiation Monitors are installed on the refueling floor to supplement the refueling floor process radiation monitors by providing radiological information to plant operators. Operators can use the vent stack and/or ARM information to manually initiate secondary containment isolation if radiological conditions warrant this action. Emergency Operating Procedures direct operator action in the event of higher than normal radiation readings.

b. The proposed change to the applicable operational condition for the railroad access shaft process radiation monitor does not affect the probability of the design basis accidents. The monitor functions in response to an airborne radioactivity concentration in the unfiltered air from the Zone III exhaust system and provides isolation signals which limit offsite doses to within regulatory limits. As such, there is no correlation between monitor operability and accident probability. The monitor acts to mitigate the offsite effects of airborne contamination producing accidents, it is not a potential accident initiator.

The proposed change does not result in a significant increase in the consequence of the design basis accidents. The design intent of the railroad access shaft process radiation monitor is to monitor radiation in the unfiltered air from the Zone III railroad access shaft exhaust system, and provide signals which automatically isolate the Zone III portion of the secondary containment, start the Standby Gas Treatment System, and start the Recirculation System (Zone III) on a high radiation condition within the access shaft. This function is intended to limit the consequences of a fuel handling accident in the railroad access shaft. The monitor has no significant capability to react to a CORE ALTERATION related transient, or one resulting from operations with the potential to drain the reactor vessel. The design intent of the monitor is maintained under the proposed change, as the proposed change focuses monitor operability on conditions when irradiated fuel is in the railroad access shaft or above it with the railroad access shaft cover open.

For the above stated reasons, the applicable operational condition for Refueling Floor Exhaust Duct High Radiation Monitors, Wall Exhaust Duct Radiation Monitors, and the Railroad Access Shaft Exhaust Duct Radiation Monitor can be modified without significantly increasing the probability or consequences of an accident previously evaluated.

II. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The Refueling Floor Exhaust Duct High Radiation Monitors, Wall Exhaust Duct Radiation Monitors, and the Railroad Access ShaftExhaust Duct Radiation Monitor function in response to an airborne radioactivity concentration in the unfiltered air from the Zone III exhaust system and provide isolation signals which limit offsite doses to within regulatory limits. As such, there is no correlation between monitor operability and the potential for creating new or different accident scenarios. The monitors act to mitigate the offsite effects of airborne contamination producing accidents, they are not potential accident initiators.

For the above stated reasons, the applicable operational condition for Refueling Floor Exhaust Duct High Radiation Monitors, Wall Exhaust Duct Radiation Monitors, and the Railroad Access Shaft Exhaust Duct Radiation Monitor can be modified without creating the possibility of a new or different kind of accident from any accident previously evaluated.

III. This change does not involve a significant reduction in a margin of safety.

a. The proposed change to the applicable operational condition for the refueling floor process radiation monitors does not involve a significant reduction in the margin of safety. The postulated event associated with control rod related CORE ALTERATIONS which could result in increased Zone III airborne radioactivity concentrations is criticality resulting from a single control rod withdrawal under shutdown conditions. There are multiple barriers to protect against the postulated event of criticality from a single rod withdrawal. Technical Specifications, plant operating procedures, and plant design control the withdrawal of control rods to minimize the potential for an inadvertent criticality event during shutdown. In addition, a fuel loading verification is performed, per procedure, on the as loaded core configuration to ensure that the fuel is loaded correctly. Each reload core is designed such that there is at least a 99.9% probability with a 95% confidence that the core will not be critical as a result of a single control rod withdrawal. The safety margin associated with a potential criticality event from a single control rod withdrawal, under shutdown conditions, is not impacted by the proposed change.

In the unlikely event that control rod manipulations resulted in reactor criticality, adequate protective measures are provided by core monitoring instrumentation required to be operable in OPCON 5. Under this scenario, assuming the inadvertent control rod withdrawal resulted in a significant reactivity addition, the Reactor Protection System (RPS) would respond by inserting all control rods via the Scram function. The RPS monitors for recriticality during OPCON 5 with SRMs (per Technical Specification Section 3.9.2), and IRMs. The safety margin associated with RPS response to a criticality event, under shutdown conditions, is not impacted by the proposed change.

Assuming that a criticality did occur as a result of a single control rod withdrawal, any increase in Zone III airborne radioactivity from a previously failed assembly located in the vicinity of the withdrawn control rod or a fuel rod failure associated with the control rod withdrawal would not result in an offsite dose exceeding regulatory limits. Assuming that criticality occurs following core loading and verification (i.e. ≤20 days after shutdown), the offsite dose as a result of the release of fission products from a single failed fuel rod would be much less than 1% of the applicable site boundary limits. In addition, the failure of four complete fuel assemblies (i.e. nearly equal to 300 fuel rods in the bundles surrounding the withdrawn control rod) would not result in offsite dose exceeding the applicable regulatory limits. Failure of more than four complete fuel assemblies due to the withdrawal of a single control rod in OPCON 5 is not considered credible. In fact, given the initial conditions of this event (i.e. cold, zero power, subcritical) and the reactivity characteristics of the fuel (i.e. negative fuel temperature reactivity coefficient) it is very unlikely that a criticality of this nature would result in failure of any fuel rods. Although the refueling floor process radiation monitors would not be OPERABLE, Zone III airborne radioactivity concentrations can be independently detected with Area Radiation Monitors (ARMs) which are located on the refueling floor. These monitors provide control room indication, and would alert operators to changing radiological conditions on the refueling floor. In addition to providing personnel notification, the ARMs act as a supplement to the process radiation monitors in detecting abnormal migrations of radioactive material in or from the process streams. Operators can manually initiate secondary containment isolation based on ARM input. The Emergency Operating Procedures require the operators to take appropriate actions on higher than normal radiation readings. Moreover, any airborne radioactivity leakage from Zone III would be monitored via instrumentation in the Reactor Building vent stack required to be OPERABLE at all times; local alarms, remote recording, and main control room and Technical Support Center alarms are provided. Operators can manually initiate secondary containment isolation based on exhaust sample readings. Due to the bounding regulatory limits and the redundant monitoring and operator response capabilities, the safety margin associated with the potential for offsite airborne radioactive release, under shutdown conditions, is not significantly impacted by the proposed change.

b. The elimination of operability requirements associated with CORE ALTERATIONS, operations with the potential to drain the reactor vessel, and other irradiated fuel moves not associated with the railroad access shaft, do not affect the ability of the railroad access shaft process radiation monitor to implement its design function. As such, the current operability requirements for the monitor which involve evolutions in areas other than the railroad access shaft do not contribute to the margin of plant safety; thus eliminating these operability requirements will not reduce the margin of plant safety.

For the above stated reasons, the applicable operational condition for Refueling Floor Exhaust Duct High Radiation Monitors, Wall Exhaust Duct Radiation Monitors, and the Railroad Access Shaft Exhaust Duct Radiation Monitor can be modified without a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: February 2, 1995

Description of amendment request: This amendment would change the Technical Specifications for the units to increase the licensed discharge fuel assembly exposure for SPC 9X9-2 fuel from 40 to 45 GWD/MTU.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not:

I. Involve a significant increase in the probability or consequences of an accident previously evaluated.

PP&L—s technical basis for increasing the licensed discharge exposure limit as proposed is documented in PL-NF-94-005-P-A. The technical basis includes onsite fuel inspections, fuel design analyses and evaluations, and an in-reactor fuel assembly extended exposure demonstration. In response to NRC concerns on fuel failures at higher exposures, very conservative analyses were performed for the CRDA [control rod drive assembly] assuming very low failure thresholds, and offsite dose calculation results were shown to be well within regulatory limits, even at a failure threshold of 30 cal/gm. The NRC has previously reviewed and approved all of the above information, and inspection results have met all approved criteria.

An evaluation of FSAR [Final Safety Analysis Report] design basis events was performed to determine the impact of the proposed increase in fuel exposure. The LOCA [loss-of-coolant accident] analysis performed in support of PP&L—s Power Uprate efforts incorporated the effects of higher exposure and LHGR [linear heat generation rate]. From a radiological release perspective, the Power Uprate evaluations of LOCA, MSLB [main steam line break], CRDA, and refueling accidents each bound the potential impacts of extended exposure fuel.

Those reload analyses deemed necessary to confirm that the above conclusions remain valid will be performed on a cycle-specific basis.

Based on the above, the proposed action will not involve a significant increase in the probability or consequences of an accident previously evaluated.

II. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed action will increase the residence time of fuel within the Susquehanna reactors. The potential consequences of this action remain solely with the fuel—s ability to perform within specified limits during the increased duty, and were reviewed in I above. All required evaluations involving fuel impacts have been previously evaluated.

Based on the above, the proposed action cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

III. Involve a significant reduction in a margin of safety.

The proposed action will allow increasing the licensed discharge fuel assembly exposure limit, resulting in increases in the fuel rod LHGR and LHGR for APRM [average power range monitor] Setpoints, which are controlled via the Technical Specifications and the Core Operating Limits Report.

The discussion in I. above delineates the evaluations performed to support this action. It concludes that neither the probability nor the consequences of events previously evaluated will be affected. Operator performance will not be affected, because the operators only monitor the ratio of the fuel LHGR to the fuel design limit. No other potentially impacted safety margins have been identified.

Based on the above, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Local Public Document Room location:*Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: February 10, 1995

Description of amendment request: The proposed amendment would modify the Susquehanna Steam Electric Station, Unit 1 and 2 Technical Specifications (TS) to (1) extend the allowable out-of-service times (AOTs) for maintenance and repair and the surveillance test intervals (STIs) between channel functional tests for the following groups of instruments: reactor protection systems instrumentation (TS 3.3.1), isolation actuation instrumentation (TS 3.3.2), emergency core cooling system actuation instrumentation (TS 3.3.3), ATWS (anticipated transient without scram) recirculation pump trip system instrumentation (TS 3.3.4.1), end-of-cycle recirculation pump trip system instrumentation (TS 3.3.4.2), reactor core isolation cooling system (RCIC) actuation instrumentation (TS 3.3.5), control rod block instrumentation (TS 3.3.6), radiation monitoring instrumentation (TS 3.3.7.1), and feedwater/main turbine trip system actuation instrumentation (TS 3.3.90); (2) change the required actions and AOTs for the instruments listed above to make requirements consistent with supporting analysis in General Electric topical reports and change additional actions required to prevent extended AOTs from resulting in extended loss of instrument function; (3) change the required actions and AOTs for the instruments listed above for instrumentation associated with the ADS (automatic depressurization system), recirculation pump trip, and pump suction lineup for HPCI (high pressure core injection) and RCIC; (4) change applicability requirements and required actions for the reactor vessel water level-low, level 3 function that isolates the RHR (residual heat removal) system shutdown cooling system so that the function is required to be operable in operational conditions 3, 4, and 5 to prevent inadvertent loss of reactor coolant via the RHR shutdown cooling system; (5) remove notes in Table 3.3.2-1, 3.3.2-2, and 4.3.1-1 related to maintenance on leak detection temperature detectors and remove the note to TS 3.3.6 for Unit 1 related to a previous relief from TS 3.0.4; and reformat, renumber, and/or reword existing requirements to incorporate the changes listed above.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes increase the AOTs and STIs for actuation instrumentation intended to detect or mitigate accidents; establish required actions consistent with NUREG-1433 for some instruments that are more specific but equivalent to existing required actions; establish new requirements to prevent inadvertent loss of reactor coolant via the RHR Shutdown Cooling System during OPERATIONAL CONDITIONS 3, 4 and 5; and, eliminate notes that were intended to provide one time only exemptions from certain requirements. The proposed changes affect only those Technical Specification requirements that govern operability, required actions and routine testing of plant instruments that detect or mitigate accidents. The proposed changes do not affect any equipment or requirements that are assumed to be initiators of any analyzed events. Therefore, the proposed changes will not involve an increase in the probability of occurrence of an accident previously evaluated.

The proposed changes will not increase the consequences of an accident previously evaluated because the changes will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested or inspected. The proposed changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. The proposed changes extend the intervals between required performances of routine instrument testing. The proposed changes also modify time limits allowed for operation with inoperable instrument channels in situations when an inoperable instrument channel would not prevent actuation of the associated equipment. These changes are based on the demonstrated reliability of these instruments and are justified by the analysis in References 1 through 8 [See February 10, 1995 application]. The small increases in the probability that the proposed changes will result in an equipment actuation failure has been determined in References 1 through 8 [See February 10, 1995 application] to be offset by safety benefits such as a reduction in the number of inadvertent actuations, a reduction in wear due to excessive testing, and better utilization of plant personnel and resources. These changes will not allow continuous plant operation with plant conditions such that a single failure will result in a loss of any safety function.

Proposed changes to required actions and completion times for instrumentation associated with the ADS initiation, Recirculation Pump Trip, and pump suction lineup for HPCI and RCIC make the required actions and completion times consistent with NUREG-1433, Standard Technical Specifications for General Electric Plants, BWR/4, Revision 0 (Reference 12). These changes are also consistent with the assumptions used in References 1 through 8 [See February 10, 1995 application]. Therefore, these changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analysis. In addition, the proposed change provides the benefit of avoiding an unnecessary shutdown transient when appropriate measures are available to compensate for the inoperable instrumentation. Therefore, the proposed changes will not increase the consequences of an accident previously evaluated.

There is no significant increase in the probability or consequences of an accident previously evaluated resulting from changes that reformat, renumber, and/or reword existing requirements to incorporate the changes above or from the removal of notes that were intended for one time only use and are no longer applicable.

II. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. The changes in normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. The proposed changes do not involve a significant reduction in a margin of safety.

The proposed TS changes: increase the AOTs and STIs for actuation instrumentation intended to detect or mitigate accidents; establish required actions consistent with NUREG-1433 for some instruments that are more specific but equivalent to existing required actions; establish new requirements to prevent inadvertent loss of reactor coolant via the RHR Shutdown Cooling System during Operational Conditions 3, 4 and 5; and, eliminate notes that were intended to provide one time only exemptions from certain requirements.

There is no significant reduction in the margin of safety resulting from changes to the minimum surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for the testing and/or repair of instrumentation. This conclusion is based on the demonstrated reliability of these instruments and is justified by the analysis in References 1 through 8 [See February 10 1995 application]. The small increases in the probability that the proposed changes will result in an equipment actuation failure has been determined in References 1 through 8 [See February 10, 1995 application] to be offset by safety benefits such as a reduction in the number of inadvertent actuations, a reduction in wear due to excessive testing.

These changes will not allow continuous plant operation with plant conditions such that a single failure will result in a loss of any safety function.

There is no significant reduction in the margin of safety resulting from changes to required actions and completion times for instrumentation associated with the ADS initiation, Recirculation Pump Trip, and pump suction lineup for HPCI and RCIC. These changes make the required actions and completion times consistent with NUREG-1433, Standard Technical Specifications for General Electric Plants, BWR/4. These changes are also consistent with the assumptions used in References 1 through 8 [See February 10, 1995 application]. Therefore, these changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analysis. In addition, the proposed change provides the benefit of avoiding an unnecessary shutdown transient when appropriate measures are available to compensate for the inoperable instrumentation. Additionally, the proposed required actions ensure that actions to mitigate loss of single failure tolerance are initiated within 24 hours (12 hours for RPS) in accordance with the results of the analyses in References 1 through 8 [See February 10, 1995 application] and action to mitigate a loss of instrument function is initiated within 1 hour. Therefore, these changes will not allow continuous plant operation with plant conditions such that a single failure will result in a loss of any safety function. The Pennsylvania Power & Light Company performed reviews that confirmed the analyses in References 1 through 8 [See February 10, 1995 application] are applicable to SSES and that there would be no effect on the identification of excessive instrument setpoint drift as a result of increasing the minimum interval between instrument functional tests from monthly to quarterly.

There is no significant reduction in the margin of safety resulting from changes that reformat, renumber, and/or reword existing requirements to incorporate the changes above or from the removal of notes that were intended for one time only use and are no longer applicable.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Public Service Electric and Gas Company,Delmarva Power and Light Company, and Atlantic City Electric Company,Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: January 13, 1995

Description of amendment request: The proposed changes concern a revision to the frequency of calibration for the Local Power Range Monitor (LPRM) signals from every 6 weeks to every 2000 Megawatt Days per Standard Ton (MWD/ST).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability of consequences of an accident previously evaluated.

This change does not affect the operation of any equipment. The change does not affect the fundamental method by which the LPRMs are calibrated. The increased time between required LPRM calibrations does not affect either the initiator of any accident previously evaluated or any equipment required to mitigate the consequences of an accident, or the isotopic inventory in the fuel. Thus, the change does not increase either the probability or the radiological consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not introduce a new mode of plant operation and does not involve the installation of any new equipment or modifications to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The GETAB determination of the Maximum Critical Power Ratio (MCPR) Safety Limit allows a maximum total nodal uncertainty of the TIP readings (of which the LPRM Update uncertainty is a part) of 8.7%. The change in LPRM calibration frequency results in an LPRM Update uncertainty of 4.2% nodal power. This, combined with the other uncertainties which comprise the total TIP readings uncertainty, yields a total TIP readings uncertainty of less than the allowed 8.7%. Thus the change in LPRM calibration frequency will not affect the MCPR Safety Limit.

The LPRMs are utilized as input to the APRM and RBM systems. The primary safety function of the APRM system is to initiate a scram during core-wide neutron flux transients before the actual core-wide neutron flux level exceeds the safety analysis design basis. This prevents fuel damage from single operator errors or equipment malfunctions. The APRMs are calibrated at least twice per week to the plant heat balance, utilize a radially and axially diverse group of LPRMs as input and are utilized to detect changes in average, not local, power changes. Therefore, the effects of decreasing the LPRM calibration frequency on the APRM system responses will be minimal due to any individual LPRM drift being practically canceled out (due to diversity of input) and/or due to the frequent recalibration of the APRMs to an independent power calculation (the heat balance). Thus, decreasing the LPRM calibration frequency will not significantly impact the performance of the APRM system's scram function, and there is no impact on transient delta-CPRs.

The RBM system is utilized in the mitigation of a Rod Withdrawal Error (RWE). The RBM system is designed to prevent the operator from increasing the local power significantly when withdrawing a control rod. On each selection of a control rod, the average of the assigned, unbypassed LPRMs is adjusted to equal a 100% reference signal for each of the two RBM channels. Each RBM channel automatically limits the local thermal margin changes by limiting the allowable change in local average neutron flux to the RBM setpoint. If the local average neutron flux change is greater than that allowed by the RBM setpoint, within either RBM channel, the rod withdrawal permissive is removed preventing further movement. Since the change in local neutron flux is calculated from the change in the average of the LPRM readings, and calibrated on every rod selection to the reference signal, offsets in individual LPRM readings due to calibration differences are effectively eliminated for a given RBM setpoint. Therefore, the constraints on the withdrawal of any given rod are unchanged and there will not be any increase in RWE delta-CPR.

Since the MCPR Safety Limit is unaffected and the delta-CPR values are unchanged, the cycle CPR limits are unchanged. Therefore, the change in the frequency of LPRM calibration does not result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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NRC Project Director: John F. Stolz

Power Authority of The State of New York, Docket No. 50-286, Indian PointNuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: February 23, 1995

Description of amendment request: The proposed amendment would change the minimum emergency diesel generator (EDG) fuel oil requirements of Technical Specifications Section 3.7 (Auxiliary Electrical Systems) from 7056 to 6721 gallons. The change is requested based on a recent modification which installed a more accurate level indicator for each of the three fuel oil tanks. The new indicators have an accuracy of plus or minus 50 gallons while the old indicators had an accuracy of plus or minus 385 gallons. Thus, the actual volume of fuel oil available to each of the EDGs remains unchanged at 6671 gallons.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed. This amendment application is the result of a modification which installed new fuel oil level indicators [for each of the three EDG fuel oil tanks]. These new indicators reduce the amount of measurement uncertainty by 335 gallons. The proposed reduction in minimum fuel oil corresponds to this reduction in uncertainty and therefore does not affect the amount of fuel oil available for use in the EDG storage tanks. This ensures that sufficient oil is present to power the minimum safeguards equipment for 48 hours, assuming two EDGs are operable.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because they do not affect the way the plant operates. This amendment application is the result of a modification which installed new fuel oil level indicators which have a higher accuracy than the previous ones. The requested change in the minimum required fuel oil volume corresponds to this reduction in measurement uncertainty. Therefore, there is no affect on the amount of oil available for use by the EDGs.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed changes do not involve a significant reduction in a margin of safety. This amendment application is the result of a modification which installed new fuel oil level indicators. These new indicators reduce the amount of measurement uncertainty by 335 gallons. The proposed reduction in minimum fuel oil corresponds to this reduction in uncertainty and therefore does not affect the amount of fuel oil available for use in the EDG storage tanks. This ensures that sufficient oil is present to power the minimum safeguards equipment for 48 hours, assuming two EDGs are operable.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request